

TREAT Transient Analysis Benchmarking for the HEU Core

Nuclear Engineering Division

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sponsored by
U. S. Department of Energy, National Nuclear Security Administration
Office of Material Management and Minimization

May 2014

Executive Summary

This work was performed to support the feasibility study on the potential conversion of the Transient Reactor Test Facility (TREAT) at Idaho National Laboratory from the use of high enriched uranium (HEU) fuel to the use of low enriched uranium (LEU) fuel. The analyses were performed by the GTRI Reactor Conversion staff at the Argonne National Laboratory (ANL).

The objective of this study was to benchmark the transient calculations against temperature-limited transients performed in the final operating HEU TREAT core configuration. The MCNP code was used to evaluate steady-state neutronics behavior, and the point kinetics code TREKIN was used to determine core power and energy during transients.

The first part of the benchmarking process was to calculate with MCNP all the neutronic parameters required by TREKIN to simulate the transients: the transient rod-bank worth, the prompt neutron generation lifetime, the temperature reactivity feedback as a function of total core energy, and the core-average temperature and peak temperature as a functions of total core energy. The results of these calculations were compared against measurements or against reported values as documented in the available TREAT reports. The heating of the fuel was simulated as an adiabatic process.

The reported values were extracted from ANL reports, intra-laboratory memos and experiment log-sheets and in some cases it was not clear if the values were based on measurements, on calculations or a combination of both. Therefore, it was decided to use the term “reported” values when referring to such data.

The methods and results from the HEU core transient analyses will be used for the potential LEU core configurations to predict the converted (LEU) core’s performance.

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1. Introduction

The Transient Reactor Test facility (TREAT) is a UO_2 -graphite fueled, graphite-moderated reactor that was designed to generate large power bursts in test samples to simulate fuel accident conditions. TREAT achieved criticality in 1959 and was operated until 1994 when it was placed on a non-operational standby status. During the operating lifetime of TREAT, hundreds of in-core irradiations subjected fueled test samples to a wide variety of power transients.

To satisfy the future needs to perform transient testing of new fuel designs, the restart of TREAT operations was approved in February 2014 [1]. For this purpose the feasibility of replacing the TREAT core with an LEU core is studied, with an objective that the new core will perform as well as the HEU core did. Because the design of the LEU core will be based on neutronics and thermal-hydraulic simulations, validated computational models and methods are required.

Steady state neutronics analyses of TREAT are currently being performed with the Monte Carlo simulation code MCNP. The 3D-detailed MCNP core model has been validated in previous studies [2] and [3]. Transient analyses of TREAT are performed with the point kinetics code TREKIN [4], which was extensively used during operation for experiment planning in the HEU core. Historically, the input data required for transient calculations with TREKIN for the HEU core was produced using a combination of measurements and theoretical calculations. For the LEU core, until measurements are available with that core, all of the analyses must be solely based on calculations. Therefore, a method which relies solely on calculations must be established and validated against measurements made in the HEU core, for use in LEU core analysis.

During the preparation of this report it was not clear if the data used to validate the calculations were based on measurements or on a combination of measurements and calculations. Therefore, it was decided to label all the data of unclear origin (measured, calculated or a combination of both) obtained through reports, memos and experiment log-sheets as “reported”.

2. TREAT Core Description

TREAT is a homogeneous, air-cooled, graphite-moderated and graphite-reflected reactor. A cut-away perspective view and top cross-sectional view of TREAT are shown in Figures 2.1 and 2.2, respectively. The reactor is fueled with UO_2 particles which are finely dispersed in a graphite matrix. It was designed, built, and operated in a manner that allowed a wide range of fueled test samples to be placed in the core, to be subject to a wide variety of power transients, all performed such that strict TREAT core-temperature limits would not be exceeded, even in the event of a reactivity accident.

A detailed description of TREAT can be found in previous reports ([2], [3]). A brief description of the core is presented in Table 2.1. The TREAT reactor cavity is designed to accommodate a maximum core size of 76-inch \times 76-inch \times 48-inch (1.9 m \times 1.9 m \times 1.22 m) high, formed by a total of 361 fuel assemblies. Table 2.1 summarizes the reactor design characteristics of key interest to the analyses discussed in this report.

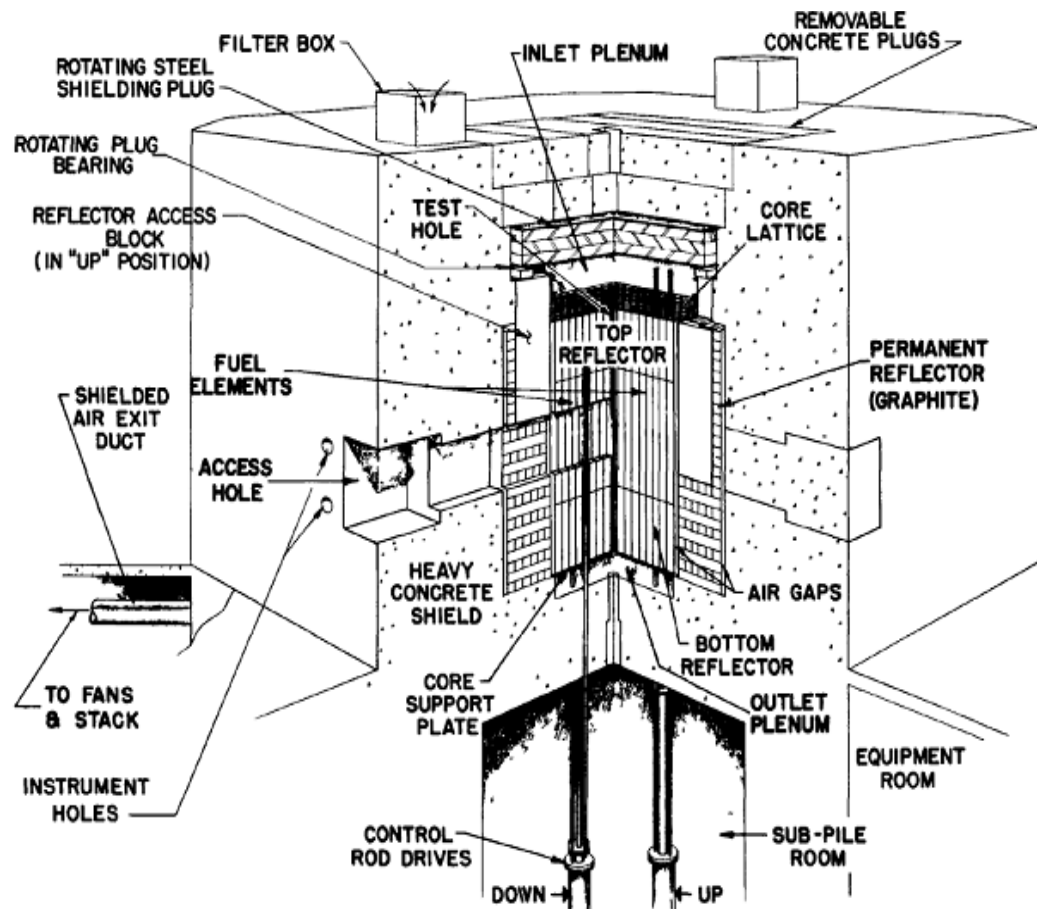


Figure [2.1]: Perspective View of TREAT

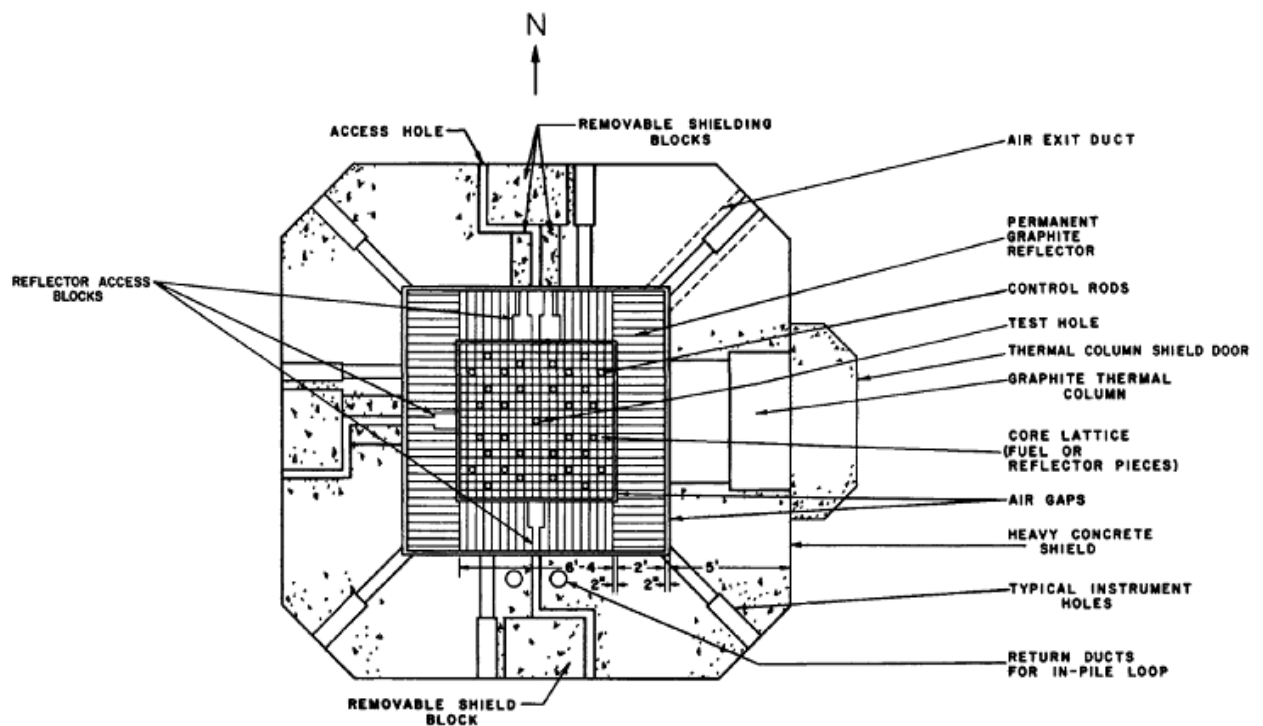


Figure [2.2]: Plan View of TREAT (the Available Control Rod Positions are Shown)

Table [2.1]: TREAT Design Characteristics

Core Parameter	Description
Maximum Core Size (361 fuel assemblies)	76 in. square, 48 in. high
Fuel Assemblies	
<i>Overall Dimensions</i>	3.96 in. square x 8.94 ft. long
<i>Fuel section</i>	Six 8-in.-long graphite-urania blocks in 25-mil thickness of Zircaloy-3 cladding, overall height 47.625 in.
<i>Total Fuel Density</i>	1.73 g/cm ³
<i>Carbon to U-235 atom ratio</i>	10,000:1
<i>U-235 enrichment</i>	93.1%
Axial Graphite Reflector	24 in. upper and lower, in 50 mils of Al cladding
Radial Graphite Reflector	24 in. thick
Coolant	Air at atmospheric pressure

An illustration of a standard TREAT fuel assembly is provided in Figure [2.3]. The four-foot-long central section of the fuel assemblies contains the fuel, an intimate mixture of enriched UO_2 and graphite, with aluminum-clad axial graphite reflectors above and below this region. In addition to the standard elements, the following special-purpose elements were used in some core configurations, including the minimum critical core:

1. Control elements: Similar to standard fueled elements, with a hole running lengthwise through which the control rods run;
2. Dummy elements: Completely filled with graphite and used to fill out the matrix around the core and provide additional reflector thickness.
3. Slotted elements: "Air filled" elements, which provide an open path between TREAT test samples at the center of the core and an ex-core set of detectors called the hodoscope.

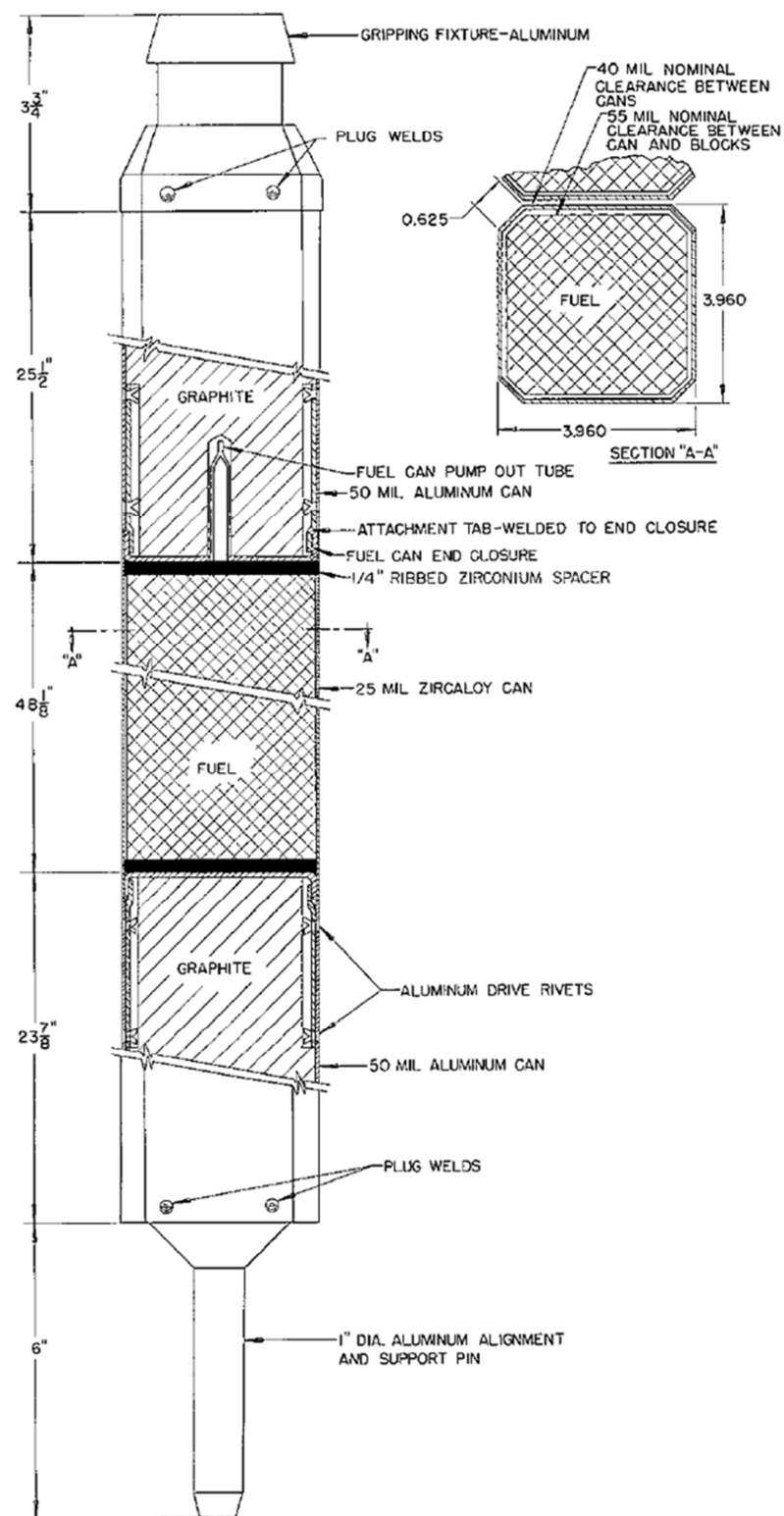


Figure [2.3]. Standard TREAT HEU Fuel Assembly

2.1 Permanent Reflector

The TREAT core is enclosed in a permanent radial reflector consisting of graphite blocks (4-inch x 4-inch x 24 inches), stacked 7 ft 8 in. high forming a 2 ft thick wall of graphite inside the reactor biological shield cavity (except for locations in the core where viewing slots into the core are provided, as described below). There is a nominal 2-inch gap on the inner and outer faces of the reflector to provide a passage for forced-air flow to cool the reactor. The reflector graphite was previously used in the CP-2 reactor.

At several locations, radial access into the core, through the biological shield and radial reflector is made possible by the presence of removable concrete shielding blocks and special large (~275 -lb) movable blocks of graphite. The blocks move vertically and are supported in the “open” position by an aluminum lifting bracket mounted on the ledge of the concrete shield above the reflector. The south and west sides of the reflector each have one movable block which provides a slot 4¼ inches wide x 24-inches high through the reflector. There are three movable blocks in the north side. The central block can be lifted to provide a slot 4 in. x 32 in., or all three blocks can be removed (through the top rotating plug) to provide an opening of 14¼ in. x 32 in. The calculations presented in this report were performed with all gates closed. (i.e., full of graphite). There are also two 6.5-in.-diameter horizontal holes in each of the north, south, and west faces of the reflector to accommodate installation of fast-response instruments close to the core.

The radial reflector is secured in position with 46 stainless steel bolts of 1 in. diameter positioned perpendicular to the core. It is supported by angle iron spacers and a sheet metal framework anchored to the steel liner of the concrete shield cavity that surrounds the reflector. As mentioned previously, there is a gap formed by the spacers and support bars, which allows for the passage of air coolant flow between the reflector and the bio-shield.

2.2 Control Rods and Driving Mechanisms

The HEU core control system features three types of reactor control rods: 1) four single pneumatic compensation/shutdown rods, used to compensate for reactivity additions; 2) four pairs of pneumatic control/shutdown rods, used to maintain criticality; and 3) four pairs of hydraulic transient rods, which provide the reactivity change to initiate TREAT transients. The in-core configuration of the three different rods banks is presented in Figure 2.4.

Transients in TREAT are performed by introducing reactivity insertion through the withdrawal of the transient rod bank. TREAT transients typically fall into one of two categories: (1) shaped transients, which can last several second and feature a very specific core power-time history is produced via slower motion of the transient rods, and (2) temperature-limited or “burst” transients, which last less than a second and are initiated by a step insertion of reactivity via the withdrawal of the transient rods at their maximum speed, causing a burst in power which is quickly constrained by the negative temperature reactivity feedback of the fuel graphite.

Figure 2.5 shows a drawing of a control rod assembly. The control rod moves within special fuel assemblies inside of a 2.25-inch-diameter hole down the middle of the fuel and axial reflector sections

of the assembly. From top to bottom, each control rod is comprised of a handling attachment, a carbon steel-clad poison section which contains boron carbide powder, a Zircaloy-clad graphite follower, and a two-piece steel-clad follower filled with graphite which connects to the drive mechanism, to form a long, cylindrical assembly (1.75 in. OD by 17ft-6½ in. long), weighing 65 lb (~30 kg). Threaded connections are installed at the ends of each section. The length of the poison section is 56.125 in.

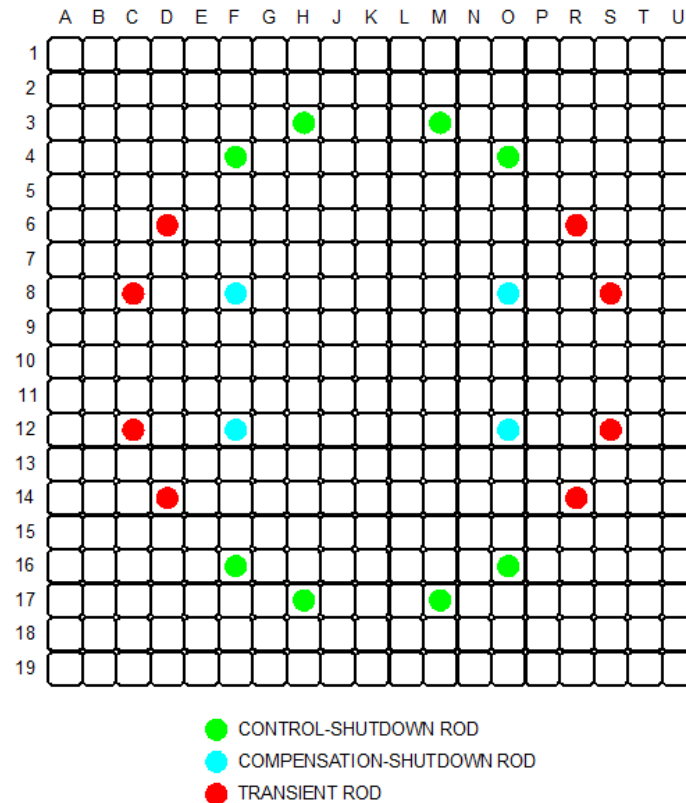


Figure [2.4]: Control Rods Position in the Studied HEU TREAT Core

2.2.1 Control Rod Components

The poison section of the TREAT control rods is a cold-drawn steel seamless tube (1.75 in. OD × 5 ft long × 1/8 in. wall) packed with boron carbide powder (B_4C) to a minimum density of 1.6 g/cm^3 [7]. The follower section is a Zircaloy-2 tube (1.75 in. OD × 5 ft long × 1/8 in. wall) filled with a stack of graphite rods. Threaded end plugs with drilled holes for the alignment studs are welded at each end of the section in close contact with the graphite filler rods. The steel follower is a carbon steel seamless tube (1.75 in. OD × 7 ft- 5 in. long × 1/8 in. wall) comprised of by two sections. The upper section (2 ft long) features an extra-long threaded male fitting which provides about 10 in. of steel shielding in the control rod thimble when the rod drive is in the “shutdown” position. The balance of the upper fitting is filled with graphite and sealed with a female threaded end plug which inter-connects with the lower steel follower section. The lower section (5 ft- 5in. long) is filled with graphite, and the end plug is threaded to receive the control rod drive knuckle fitting.

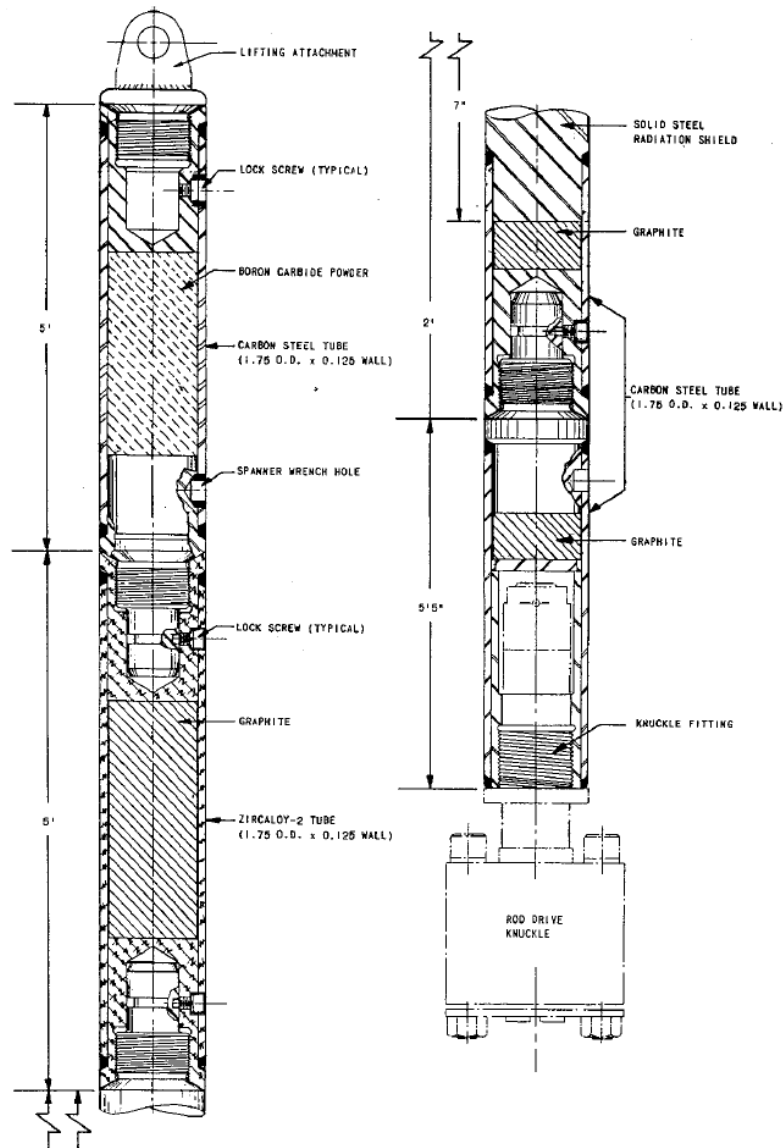


Figure [2.5]: Control Rod Drawing Showing the Different Segments

All three rod banks can be used for scram shutdowns initiated by the reactor trip system, and all rod banks are designed to a single-failure criterion. The transient, control/shutdown and compensation/shutdown banks are each, independently, capable of providing cold shutdown. The transient drives are hydraulically actuated and have a 40 inch range. When the reactor is operating under computer control (i.e., transient operation), the transient rods are allowed to add reactivity at various speeds up to their maximum speeds (130-160 in./s) as necessary to produce the required reactor power-time history. During all other times, the high-speed capability is locked out, and the rods can move at only 0.67 in/s.

The four control/shutdown rod drives and the four compensation/shutdown rod drives each use a lead screw with a 60-in. stroke and employ a pneumatically-assisted scram. Because they are limited to a maximum withdrawal speed of 19.75 in./min, they are used for manual control during steady-state reactor operation.

The compensation/shutdown rod drives have the same design as those used for the control/shutdown rods. However, the compensation/shutdown rod drives use a high-speed hydraulic latch to achieve a faster scram initiation. The compensation/shutdown rods are typically fully out of the core at the beginning of any reactor operation and are fully inserted prior to any experiment removal from the core. Their purpose is to compensate for the reactivity addition which occurs from experiment removal.

3. TREAT Fuel Composition

As discussed above, the TREAT fuel consists of UO_2 particles dispersed in graphite. Its exact composition is not certain. One of the most significant uncertainties in the composition is the fuel boron impurity level, which is significantly greater than that of typical reactor-grade graphite due to the accidental introduction of boron during the manufacturing process. In addition, the fuel carbon is believed to be only partially, rather than fully, graphitized, as discussed in previous studies [5]. In 1970 Brittan and Doerner addressed the uncertainties and proposed reference values to be used for future physics calculations. This data set was used for the neutronics presented in this report. The U-235 density in TREAT fuel was calculated by Brittan and Doerner using the inventory data from the records kept by the Argonne Special Materials group (shown in Table 3.1).

Table [3.1]: Reference TREAT Fuel Composition Used in MCNP

Fuel Element	Number	U-235 g /element	Element Volume (cm ³)
Standard	302	37.42115	11036.542
Thermocouple	29	37.064137	n/a*
Slotted	21	18.64194	5503.788
Control	30	25.987	7708.884

* the volume of the Thermocouple fuel element was not reported

Since the fuel volume of an element with thermocouple holes was not available, Brittan and Doerner calculated the average U-235 density as follows:

$$\left(\frac{302 \times 37.42115}{11036.542} + \frac{21 \times 18.64194}{5503.788} + \frac{30 \times 25.987}{7708.884} \right) / (302 + 21 + 30) = 3.3887 \times 10^{-3} \text{ g/cm}^3 \text{ of U-235}$$

4. TREKIN Code

TREKIN [4] is a point kinetics/transient simulation code, written specifically for application to the TREAT reactor that relates the reactivity insertion to the reactor power and total energy release. TREKIN utilizes a set of predetermined core specific neutronics parameters. Historically these parameters were determined via in-core measurements. The focus of this work was to validate a method which relies on calculations for these parameters, since measurements are not available for the LEU core.

TREKIN solves the kinetics equation using six delayed neutron energy groups in either of two modes; “rod/reactivity-driven” or “power-driven”, with input being the time-dependent rod position (or

reactivity insertion) or the time-dependent power, respectively. In the rod/reactivity-driven mode, TREKIN uses user-provided inserted reactivity vs. time values (input “data”) to compute the resultant power-time history, while in the power-driven mode, TREKIN uses a user-specified power vs. time dataset to calculate the reactivity insertion vs. time (i.e., rod position vs. time) needed to produce the desired power-time history. The code requires the following input data; the label of the data card used in the TREKIN input is mentioned in the parentheses:

- a. Physical parameters of the core
 - a. Available transient rod reactivity as a function of rod bank withdrawal (TROD)
 - b. Effective delayed neutron fraction (β_{eff}), prompt neutron generation lifetime (l_p) and six delayed neutron energy group structure (PHYSICS CONSTANTS)
 - c. Maximum core (hot-spot) temperature as a function of TREAT energy (TEMPER)
 - d. Temperature reactivity feedback as a function of TREAT energy (HSLOT)
- b. Operating parameters of a particular transient
 - a. Power time history for power-driven mode (TD table), or
 - b. Transient rod position time history, or Input reactivity time history, for rod driven mode (TD table)

Historically, TREAT has been considered to be a relatively tightly-coupled reactor, so it is unlikely that space-dependent kinetics will play an important role.

5. Safety Limitations

The TREAT reactor operational safety limits (SL) are defined as the values of reactivity, power and energy that would cause the maximum fuel assembly temperature to reach 820°C. The TREAT limiting safety system settings (LSSS) are defined to be the limits that would prevent the maximum fuel assembly temperature from exceeding 600°C during normal operation. These temperature limits were in place to prevent excessive oxidation or material structural changes of the TREAT fuel cladding. The measurements performed to determine the SL and the LSSS for the original HEU core were used as benchmarks for the MCNP/TREKIN transient calculations.

In order to assure that the LSSS is not exceeded, a requirement was placed on experiments such that planned transients will not produce maximum core fuel temperatures above 575°C (including an extra 25°C margin).

In order to assure that the SL was not exceeded, the excess reactivity allowed in the transient rods was limited such that the maximum core fuel temperature would not exceed 820°C if the total excess reactivity available in transient-type control rods, plus credible positive reactivity feedback effects that could be caused by experiment response to the power transient, were to be accidentally added as a step with the power transient terminated by the core’s negative temperature coefficient only (i.e., no reactor scram from any control rod bank).

TREAT reactor trip points were set to prevent exceeding the LSSS during normal operation and the SL in an accident scenario. These points were unique for every combination of core loading and

experiment vehicle. A method for determining the necessary limits on reactor period, energy release, power, and available reactivity was developed by the TREAT operators. This method required as input a set of measured data from a series of experimentally performed temperature-limited transients, for increasing step reactivity insertion values (all of which produced temperatures below the LLSS). According to the description provided in the TREAT SAR the procedure used by the reactor operators to evaluate the appropriate limits for the reactor trip points was as follows:

1. Calculate the temperature reactivity feedback $(\Delta\rho)/(\Delta T)$ as a function of the average core temperature using point kinetics theory:

$$(\Delta\rho) = \frac{\Delta k_{eff}}{k_{eff}} = ((S_{a2}^P + D_s B^2) - \frac{\left(\frac{S_{a2}^{25}}{S_{a1}^{25}}\right)(S_{a1}^P + D_s B^2)}{S_{a2} + D_s B^2})$$

where

S_a^P = Average macroscopic parasitic absorption cross – section

S_a^{25} = Average macroscopic ^{235}U fission cross – section

S_a = Average macroscopic total absorption cross
– section, including fission for a Maxwellian distribution

D_s = Scattering diffusion length of carbon

B^2 = Buckling

and

Subscripts 1 & 2 refer to initial and final temperatures, respectively

2. Calculate the temperature reactivity feedback $(\Delta\rho)/(\Delta T)$ as a function of the peak core temperature
 - a. Develop a curve of peak-to-average core temperature ratios as a function of average core temperature, using an iterative method as follows:
 - i. Using an initial value of peak-to-average power of 1.70, along with the enthalpy curve for the TREAT fuel and the temperature-dependent fission cross sections for U-235, and assuming the initial peak-to-average power ratio is constant over a small increase in peak core temperature, determine the average core temperature from the enthalpy curve and the peak-to-average power ratio.
 - ii. Then, correct the peak-to-average power ratio using the ratios of the hot U-235 fission cross section to the cold U-235 fission cross section for the peak and the average power. Repeat the procedure using the newly-corrected peak-to-average power value. (The peak-to-average ratio soon converges to 1.4; refer to TREAT SAR for more details).
 - b. Combining the result of this step with the result of step 1 yields the temperature reactivity feedback as a function of peak core temperature.
3. Construct a plot of initiating reactivity vs feedback reactivity using the measured values of initiating reactivity from experimentally-performed temperature-limited transients and the corresponding values of feedback reactivity determined in step 2. The Nordheim-Fuchs model [14] of a point reactor predicts that for large increases in reactivity, the initial reactivity and the

- total feedback reactivity exhibit a linear relationship. This linear relationship changes from core to core but is unique for a given core.
4. Using the 600°C and 820°C feedback reactivity values from the table of feedback reactivity vs peak core temperature and the plot of initiating reactivity limits vs feedback reactivity, determine values of initiating reactivity which correlate to 600°C and 820°C peak temperature for the new core of interest.

A similar method was used in this report, but the neutronics analyses were based on the 3D-detailed MCNP model of the TREAT core and the point kinetics code TREKIN. The calculation steps are described in the following sections.

6. MCNP Calculations

MCNP is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon or coupled neutron/photon/electron transport, including the capability to calculate eigenvalues for critical systems. The code uses pointwise cross-section data for all the particle interactions, and the thermal neutrons are described by both free-gas and $S(\alpha, \beta)$ models [12].

An MCNP model of the TREAT HEU half-slotted M8CAL core [6] was used in the calculations described in this report. This core consisted of 318 standard fuel assemblies, 20 control-rod fuel assemblies, 12 graphite dummy assemblies in the corners of the core, 8 slotted (air-filled) assemblies, the M8CAL vehicle (located at the center and occupying the space of two fuel assemblies), and one half-size graphite dummy assembly and one half-size slotted assembly on each side of the M8CAL vehicle (Figure 6.1). The level of graphitization (ratio of graphite to total carbon) used in the model was 59%.

The data required for the TREKIN calculations of transients was produced with MCNP. These calculations involved: (a) transient rod bank worth, (b) temperature reactivity as a function of core energy, (c) peak and average temperatures as functions of core energy and (d) effective delayed-neutron fraction β_{eff} and prompt neutron generation lifetime (l_p).

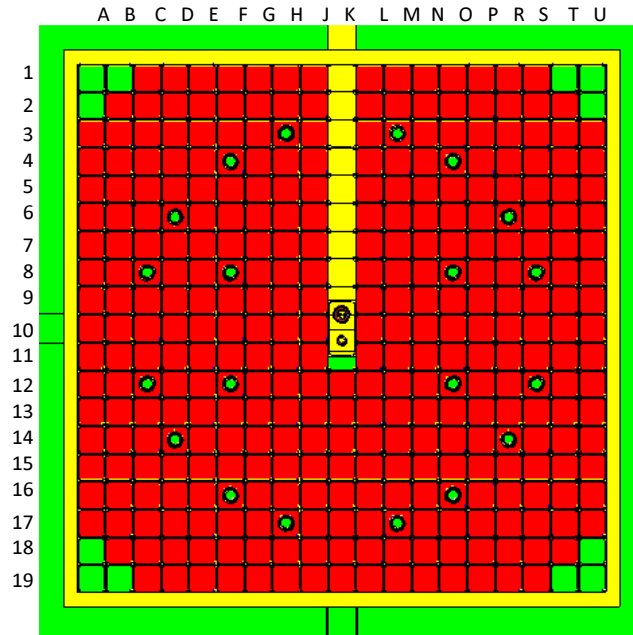


Figure [6.1]: MCNP Model of the Half Slotted HEU TREAT core. The M8CAL Vehicle in Core Center

■ Graphite Reflector, ■ Fuel Assembly, ● Control Assembly, ■ Air

6.1 Transient Rod Bank Worth

The worth of the transient rod bank was calculated by withdrawing the rods in 5 in. steps from a fully-inserted position. The control/shutdown rod bank was incrementally inserted at each step to approximately achieve criticality for every transient rod bank position (level of insertion). The worth of the inserted length of the transient rods was calculated as the reactivity change from the critical state to the transient-rods-out configuration. Table 6.1 shows the initial insertion of the control/shutdown and transient rods and the resulting reactivity change (worth) after fully withdrawing the transient rods. The standard deviation in the k_{eff} values from the MCNP calculation was 0.00019.

Table [6.1]: Calculated Transient Rod Bank Worth

Initial Rod Bank Insertion		k_{eff}		Transient Rod Bank Worth (dk/k)
Transient	Control/Shutdown	Trans In	Trans Out	
100.0%	0.0%	0.98459	1.08143	9.09%
87.5%	0.0%	0.99450	1.08143	8.08%
75.0%	8.6%	1.00671	1.07881	6.64%
62.5%	29.3%	1.00614	1.06169	5.20%
50.0%	40.3%	1.00676	1.04574	3.70%
37.5%	49.0%	1.00735	1.03090	2.27%
25.0%	56.4%	1.00623	1.01853	1.20%
12.5%	60.3%	1.00771	1.01235	0.45%

The calculated rod worth was compared with the reported values (no uncertainty was reported). It should be noted that in the MCNP model the density of the B_4C -filled section was assumed to be 1.6 g/cm³ which, according to ANL-6034 [10], is the minimum acceptable density. The ratio of the

reported-to-the MCNP-calculated worth was found to be almost constant at a value of 0.92 (with a standard deviation of 0.02), so it was decided to adjust the calculated values by multiplying them by 0.92 when used by TREKIN to determine the transient rod position during shaped transient calculations. This adjustment only affects the reported rod position (in inches) and does not affect any other TREKIN calculation. It should also be noted that the B₄C density of the rods was reported with values ranging from 1.4 to 1.8 g/cm³. However it was calculated that after perturbing the density by ± 0.2 g/cm³, the total worth of the transient rods changed by less than 0.1%. Table 6.2 shows the calculated and the adjusted values in comparison with the measurements.

Table [6.2]: Calculated, Adjusted and Reported Transient Rod Bank Worth

Transient Rod Bank Withdrawal (%)	Rod Bank Worth (dk/k)		
	MCNP	Adjusted MCNP	Reported
100.0%	9.09%	8.37%	8.46%
87.5%	8.08%	7.44%	7.49%
75.0%	6.64%	6.11%	6.24%
62.5%	5.20%	4.78%	4.84%
50.0%	3.70%	3.41%	3.42%
37.5%	2.27%	2.09%	2.13%
25.0%	1.20%	1.10%	1.10%
12.5%	0.45%	0.42%	0.40%
100.0%	0.00%	0.00%	0.00%

6.2 Effective Delayed Neutron Fraction (b_{eff}) and Prompt Neutron Generation Lifetime (l_p)

The effective delayed neutron fraction (b_{eff}) and the prompt neutron generation lifetime (l_p) were calculated with MCNP using the 'kopts' card. The prompt neutron generation lifetime (l_p) was calculated with a value of 8.6×10^{-4} s and the reported value for different core loadings was 9.0×10^{-4} s. Table 6.3 shows the MCNP calculated and the reported (ANL-6174 [7]) six delayed neutron groups and the resulting b_{eff} . It should be noted that no uncertainty has been reported for these values.

Table [6.3]: Calculated and Reported Delayed Neutron Parameters and Prompt Neutron Generation Lifetime

Precursor	Decay Constant (sec ⁻¹)		$b_{\text{eff}, i}$	
	MCNP (ENDF/B-VII)	Reported	MCNP (ENDF/B-VII)	Reported
1	0.01249	0.0124	0.00022	0.000244
2	0.03182	0.0305	0.00114	0.001567
3	0.10938	0.1110	0.00111	0.001411
4	0.31699	0.3010	0.00311	0.002828
5	1.35398	1.1300	0.00091	0.000826
6	8.63652	3.0000	0.00032	0.000302
		b_{eff}	0.00680	0.007178

The differences observed in the decay constant values for the fourth, fifth and sixth delayed neutron groups do not have a significant effect in the calculation of the peak power and maximum temperature in a temperature-limited transient.

6.3 Core Average and Hot Spot Temperatures

The SL and LSSS values for the available reactivity before a planned experiment were historically determined by extrapolation of the peak core temperature and inserted reactivity measured during three experimentally-performed temperature-limited transients (for M8CAL, these were transients #2855, #2856 and #2857). Typically for these transients, the transient and control/shutdown rods were set in a pre-transient configuration (critical state) that was determined in part by the reactivity (available reactivity in the transient rods) required for the given experiment. The transient rods were then withdrawn as a bank at maximum speed, generating a power pulse that was limited by the temperature reactivity feedback, which returns the power to a low level.

During the temperature-limited transients, the heat deposition in the TREAT fuel happens very quickly (the peak power is reached in less than a second) before significant heat loss from the fuel occurs, so it can be simulated as an adiabatic process. The core power generated, and resultant temperature distribution, depends on the reactivity insertion during an experiment. Since there will be no measurements for all the potential LEU core designs until an LEU core is installed, a method to evaluate the core temperatures solely based on calculations was developed. The temperature calculations utilize the relative power distribution calculated for the final rod configuration, i.e., with the transient rods 100% withdrawn from the core and the control/shutdown rods in the position which held the core at critical prior to the transient. Therefore, the power distribution depends only on the axial position of the control/shutdown rods.

From the three temperature-limited transients for M8CAL, the one with smallest transient-rod-bank reactivity insertion (transient #2855) was chosen for use in generating the TREKIN input data. This was done because this transient was the one with the greatest insertion of the control/shutdown rods. This was expected to result in higher peak assembly temperatures for the same total core energy because the position of the rods tilts the neutron flux towards the core center. The pre-transient rod configuration for transient #2855 had transient rods 66.25% withdrawn, control/shutdown 49.53% withdrawn, and compensation/shutdown rods 100% withdrawn. After changing the transient rod positions to 100% withdrawn, the relative power distribution was calculated with MCNP (using F6 tallies in neutron/photon mode assuming 193.7MeV/fission [18]) for a cold core (cross sections set at room temperature). Using the measured fuel-graphite heat capacity as a function of temperature [13] and the MCNP-calculated relative power distribution, the average temperature of every fuel assembly was calculated for total core energy steps of 100, 200, 300, 400, 500, 1000, 2000, 3000, 4000 and 5000 MJ. It should be noted that the relative power distribution for the “initial” zero MJ core energy (core at room temperature) was determined using the cross section libraries at 20.45°C. It was calculated that switching the cross sections from 20.45°C to 26°C (core average temperature) caused a change in k_{eff} of less than 0.2%.

For each energy step and resultant core temperature distribution, the core was simulated in its “hot” condition in MCNP, i.e., simulated using cross-sections and physics parameters $[S(\alpha, \beta)]$ at elevated

temperatures. For these simulations, the core was divided into temperature zones, each of which was assigned a representative temperature. The fuel section of every fuel assembly was divided into three axial segments. The temperature of each axial segment was calculated assuming that the axial relative power distribution in every assembly to be equal with the core average axial power distribution. The collection of fuel assemblies was divided into three groups (roughly speaking, radial “zones”) following the clustering method as described by Aravind, et al. [9]. The assembly temperature in each group was then assigned a temperature equal to the average temperature of all of the fuel assemblies in that group. The TREAT core temperature distribution was thus approximated by three radial and three axial temperature zones in each hot core simulation. See Appendix B for more information. The hottest fuel assembly was taken to be the fuel assembly with the highest average temperature. The axial hot spot temperature was calculated for each energy step using the MCNP-calculated axial relative power distribution (evaluated using fifteen axial segments) for the fuel assembly in question. The resulting calculated core-peak and average fuel temperatures as functions of core energy were used as input for the TREKIN calculations. The reported pre-transient core-average temperature ranges from 21 to 26°C. The room temperature for the MCNP cross section was set at 20.45°C. Considering the less than 0.2% k_{eff} decrease between 20.45 to 26°C average core temperature, and for comparison purposes with the data available in TREKIN, the core-average and hot-spot temperatures were calculated starting from 26°C core average temperature. Table 6.4 shows (a) the MCNP-based calculations of the hot-spot and core-average temperatures as functions of the total core energy and (b) the reported values that were included in the current version of TREKIN. The differences observed between the MCNP-calculated and historic core average and peak fuel temperatures may be due to graphite’s heat capacity (there is no information on the heat capacity used to obtain the historic data) and the relative power distribution used in both cases. Appendix A contains the historic and the MCNP-calculated tables as used by TREKIN.

Table [6.4]: Calculated and Reported Core Average and Peak Temperatures

Total Core Energy (MJ)	Core-Average Fuel Temperature (°C)			Peak Fuel Temperature (°C)		
	<i>Calculated (C)</i>	<i>Reported (R)</i>	<i>C/R</i>	<i>Calculated (C)</i>	<i>Reported (R)</i>	<i>C/R</i>
0	26	26	1.00	26	28	0.93
100	46	44	1.05	61	61	1.00
200	65	64	1.02	92	94	0.98
300	82	83	0.99	120	125	0.96
400	99	100	0.99	146	153	0.95
500	114	118	0.97	170	180	0.94
1000	184	192	0.96	277	296	0.93
2000	299	316	0.95	452	488	0.92
3000	398	424	0.94	605	659	0.92
4000	489	524	0.93	747	821	0.91
5000	574	618	0.93	882	974	0.91

The calculated values are lower than the reported core-average and peak fuel temperatures by up to 7% and 9%, respectively. During the preparation of this report no information regarding the KENO-V calculations (the resource for the ‘reported’ values) was available, so the differences with the MCNP calculations could not be further investigated.

According to R W Swanson [17], the Monte Carlo code KENO-V was historically used to calculate the temperature reactivity feedback; however, no details on the applied methodology were provided in that document. The peak-to-average power and the core average J/g/MJ calculated with KENO-V and MCNP are shown in Table 6.5. The core average J/g/MJ was calculated with MCNP assuming 193.7 MeV released per fission. No information about the energy released per fission value assumed in the KENO calculations was available.

Table [6.5]: Radial Peak-to-Average Power, Axial Peak-to-Average Power Core and Average J/g/MJ Calculated with MCNP and KENO-V

Parameter	MCNP (this study)	KENO-V (Swanson)	MCNP/KENO-V
Radial Peak-to-Average Power	1.48	1.46	1.01
Axial Peak-to-Average Power	1.22	1.25	0.98
Core Average J/g/MJ	0.15081	0.15855	0.95

6.4 Temperature Reactivity Feedback

For each of the three radial and three axial temperature zones, a different cross-section set was assigned. The temperature-dependent cross sections were derived by extrapolating the existing MCNP cross sections (ENDF/B-VII) using the MCNP utility program *makxsf* [15].

The change of reactivity as a function of total core energy was calculated as the change of the MCNP-calculated effective multiplication factor when using the cross sections at room temperature relative to using cross sections at the resulting temperature. Table 6.6 shows the MCNP-calculated and reported absolute reactivity change as functions of total core energy. The peak fuel temperatures from Table 6.5 are also shown along with the calculated-to-reported ratio values.

The difference observed between the calculated and reported temperature reactivity may be due to the fuel-graphite heat capacity (the heat capacity data that was used historically to generate the reported reactivity is not known) and/or the relative power distribution used to calculate the temperature distribution. Comparing the peak fuel temperature values, it appears that the peaking factors assumed in MCNP are lower than those in the reported data.

Table [6.6]: Calculated and Reported Reactivity Change as a Function of Total Core Energy and Peak Fuel Temperature

Total Core Energy (MJ)	Reactivity Change ($1/k_{HOT} - 1/k_{COLD}$)			Peak Fuel Temp (°C)		
	Calculated (C)	Reported (R)	C/R	Calculated (C)	Reported (R)	C/R
0	0.00%	0.00%	-	26	28	0.93
100	0.60%	0.55%	1.09	61	61	1.00
200	0.97%	1.07%	0.91	92	94	0.98
300	1.35%	1.56%	0.87	120	125	0.96
400	1.77%	1.98%	0.89	146	153	0.95
500	2.13%	2.38%	0.90	170	180	0.94
1000	3.63%	4.03%	0.90	277	296	0.93
2000	5.87%	6.35%	0.92	452	488	0.92
3000	7.78%	8.02%	0.97	605	659	0.92
4000	9.29%	9.40%	0.99	747	821	0.91
5000	10.60%	10.55%	1.00	882	974	0.91

According to Knapp [8], differences between experimentally-measured and TREKIN-calculated transient power-time histories occurred due to the experimental definition of Mega-Joules in TREAT applications and the way that the flux detectors were calibrated to measure core power. The flux detectors were located outside of the core, so their response (i.e., relationship between number of fissions in the core and flux at the detector) changed not only with core temperature (as the core heats up, the neutron leakage increases, and more neutrons reach the detectors) but also with control rod insertion (as the control rods are withdrawn, the neutron flux at the detectors increases).

The calibration of these detectors was performed once for each core loading, for a single set of core conditions, so the measured power was not truly representative of the power generated in the fuel as core conditions changed during a transient. During TREAT operations a constant, multiplicative, feedback factor (referred to as the 'SLOTK' variable) was incorporated in TREKIN to adjust the temperature reactivity feedback vs. core energy values so that the calculations would better match the measurements for each core.

7. TREKIN Calculations

The input tables required by TREKIN were calculated with MCNP for the simulation of temperature-limited transients and shaped transients. The measured temperature-limited transients that were experimentally performed in the half-slotted TREAT core for the determination of the SL and LSSS set points (see section 5) for the M8CAL half-slotted core were used as benchmarks for the TREKIN calculations using the MCNP-produced input tables. The TREKIN calculations were also performed using the already-available tables (labeled as 'historic') for comparison purposes. Appendix A shows the historic and MCNP-calculated tables used by TREKIN.

One shaped transient (#2874) was also included among the TREKIN calculations. For that analysis, the assumption was made that the relative power distribution remains constant during the entire transient regardless of the rod movement.

7.1 Temperature Limited Transients

Three temperature-limited transients (nos. 2855, 2856 and 2857) were experimentally performed in the half-slotted HEU core (loading no. 6541) in 1992. Table 7.1 shows the measured parameters as reported in the TREAT “Safety Limits and LSSS Set Points Worksheet” for those transients. The inserted reactivity values reported in those worksheets are slightly different from the ones reported in the transient log sheet; values from the latter are shown inside parentheses in Table 7.1. It should be noted that the measurement uncertainties were not reported so they were not available during the preparation of this report.

Table [7.1]: Experimentally-Performed Temperature-Limited Transients in Performed in the HEU M8CAL Half-Slotted Core

Transient Number	Inserted Reactivity	Peak Fuel Temp (°C)	Feedback Reactivity	Peak Power (MW)	Core Energy (MJ)
2855	1.81% (1.80%)	236	3.246%	1281	792
2856	3.02% (3.00%)	378	5.150%	6171	1572
2857	3.87% (3.85%)	488	6.441%	12493	2265

The TREKIN calculations were performed using both the historic tables (which were used with TREKIN at the time those transients were performed) and the MCNP-calculated TREKIN input tables. Tables 7.2 and 7.3 show the results of those calculations. For comparison purposes the SLOT factor was set at unity for all the cases. The “reported” values in both tables are the measured values indicated in Table 7.1.

Table [7.2] TREKIN Results for the Temperature-Limited Transients Using the Historic Tables

Transient Number	Inserted Reactivity	Peak Fuel Temp (°C)		Peak Power (MW)		Core Energy (MJ)	
		<i>Reported</i>	<i>Calculated</i>	<i>Reported</i>	<i>Calculated</i>	<i>Reported</i>	<i>Calculated</i>
2855	1.81%	236	237	1281	1291	792	737
2856	3.02%	378	384	6171	5781	1572	1442
2857	3.87%	488	498	12493	11398	2265	2055

Table [7.3] TREKIN Results for the Temperature-Limited Transients Using the MCNP Tables

Transient	Inserted Reactivity	Peak Fuel Temp (°C)		Peak Power (MW)		Core Energy (MJ)	
		<i>Reported</i>	<i>Calculated</i>	<i>Reported</i>	<i>Calculated</i>	<i>Reported</i>	<i>Calculated</i>
2855	1.81%	236	245	1281	1521	792	853
2856	3.02%	378	398	6171	7102	1572	1689
2857	3.87%	488	511	12493	13849	2265	2382

In order to compare the TREKIN results using the two different sets of feedback tables, calculated-to-reported ratios were evaluated for the peak temperature, peak power and core energy values. This was done for calculated results using both the historic and MCNP-calculated TREKIN tables. Table 7.4 summarizes the comparison of the resulting ratios.

Table [7.4]: Calculated-to-Reported Ratio for the TREKIN Calculated Peak Temp, Peak Power and Core Energy Based on Historic and MCNP Tables

Transient	Inserted Reactivity	Calculated-to-Reported Ratio					
		Peak Temp		Peak Power		Core Energy	
		Historic	MCNP	Historic	MCNP	Historic	MCNP
2855	1.81%	1.00	1.04	1.01	1.19	0.93	1.08
2856	3.02%	1.02	1.05	0.94	1.15	0.92	1.07
2857	3.87%	1.02	1.05	0.91	1.11	0.91	1.05

The TREKIN results produced using the MCNP-calculated tables overestimate the peak temperature, the peak power and the total core energy for the three transients, with a calculated-to-reported ratios ranging from 1.04 to 1.19. These differences are attributed to the MCNP-calculated temperature reactivity feedback (when expressed as a function of core energy) being lower by 3% to 13% from the reported ones for core energies between 0 to 3000 MJ. By choosing the appropriate value for the SLOT factor the ratios are improved, as shown in Table 7.5 where SLOT values were used to match the calculated and reported peak temperatures.

Table [7.5]: Calculated-to-Reported Ratio for the TREKIN Calculated Peak Temp, Peak Power and Core Energy Based on MCNP Tables Using the SLOT Factor

Transient	Inserted Reactivity	SLOT	Calculated-to-Reported Ratio		
			Peak Temp	Peak Power	Core Energy
2855	1.81%	1.04	1.00	1.13	1.02
2856	3.02%	1.05	1.00	1.08	1.00
2857	3.87%	1.05	1.00	1.05	0.98

7.2 Shaped Transient TREKIN Calculations

During TREAT operations and before performing a planned experiment TREKIN was used to estimate the rod-position vs. time history that would produce the required power-vs.-time history for a planned experiment. In this study, shaped transient #2874 was chosen as a benchmark for TREKIN calculations. Transient #2874 was the transient with the highest total core energy performed in the M8CAL experiment series. It was generated by a power rise on two successive periods: 0.3 and 8.0 seconds. Power-driven and rod-driven TREKIN simulations were performed using the historical and the MCNP-produced input tables, and the calculated rod and power time-histories were compared with the measured ones. In the TREKIN calculations the SLOT factor was set to unity (no adjustment in the temperature reactivity feedback).

7.2.1 Rod-Driven Calculation

The measured time-history of the transient bank position of transient #2874 was used to drive the TREKIN calculations using the historic and the MCNP-produced input tables. The simulation was period-driven (with a period of 0.3s) during the first 2.8s before switching to a power-driven mode until the end of the transient. The reason for using a period-driven segment is the artifact power peak appearing right after the heating up period (during the first 2.8s) when the transient calculation was completely power-driven [4].

The resulting power and energy time-histories were compared with the measured ones. Figure 7.1 shows the measured and calculated power time-histories and Figure 7.2 the measured and calculated energy time-histories.

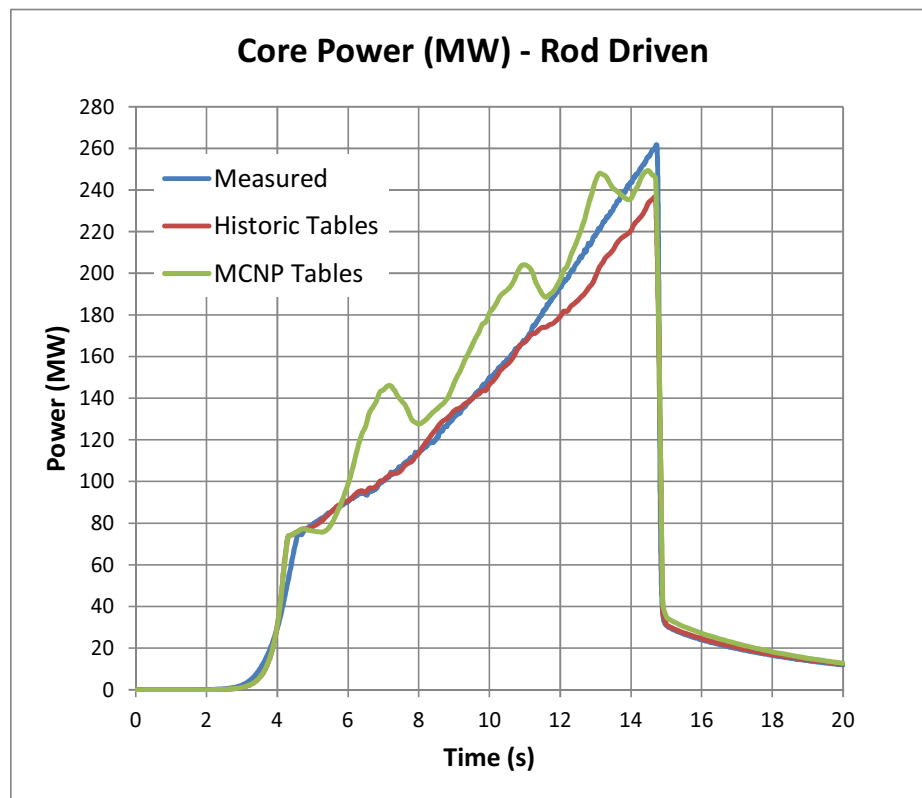


Figure [7.1]: Measured and TREKIN-Calculated (Rod Driven) Core Power Time Histories for Shaped Transient #2874

As it is described in the TREKIN user's guide [4], the rod-motion fluctuations result in sudden up or down changes to the exponential period of computed power. TREKIN uses the table of the transient rod worth as a function of rod withdrawal to convert the rod position time-history (which drives the calculation of power) into reactivity time-history. Therefore the computed power also depends on the rod worth data used. Recall that the MCNP-calculated transient rod worth values were found to be higher than the reported ones [2], so they were adjusted by multiplying each rod worth value by 0.92. The magnitude of the artifact power peaks also depends on the temperature reactivity feedback because power peaks are inversely proportional to the magnitude of the feedback (lower feedback causes higher the power peaks for the same rod-motion fluctuations). A sensitivity analyses is planned

to investigate the creation of these power artifacts. This will include further evaluation of the errors in the rod worth calculations.

Figure 7.2 shows the measured and calculated total core energy as a function of time. TREKIN calculates the total core energy by integrating power, and the result is not as sensitive to the rod position fluctuations so there are no artifact spikes.

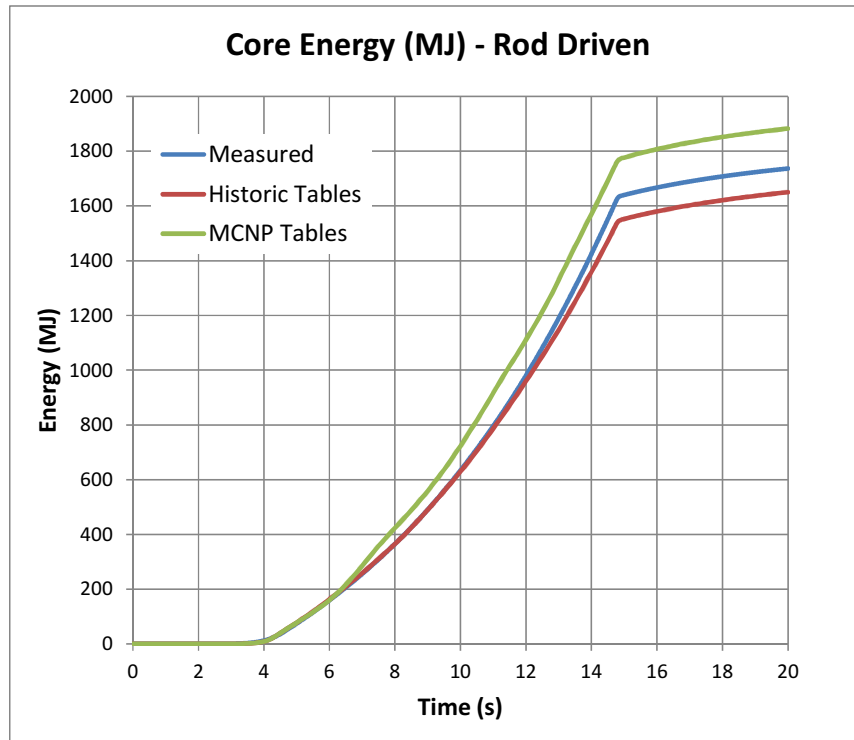


Figure [7.2]: Measured and TREKIN-Calculated (Rod Driven) Core Energy Time History for the Shaped Transient #2874

7.2.2 Power-Driven Calculation

The measured power time-history of the shaped transient #2874 was also used to drive the TREKIN calculations (power-driven mode) using the historic and the MCNP-produced tables.

The measured and the TREKIN-calculated rod time-histories are shown in Figure 7.3. Using the MCNP-produced tables, TREKIN underestimated the rod worth needed to produce the given power time-history, due to the lower temperature reactivity feedback calculated with MCNP (see Table 6.6). The lower temperature reactivity feedback means that there is lower 'demand' for reactivity insertion and hence rod withdrawal.

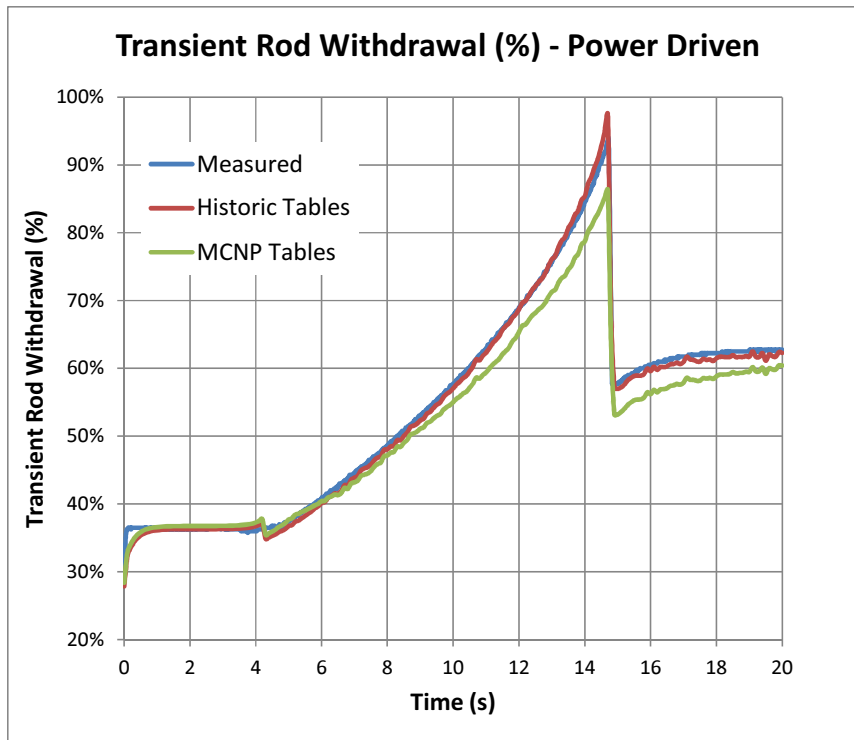


Figure [7.3]: Measured and TREKIN-Calculated (Power Driven Mode) Core Rod Withdrawal Time History for the Shaped Transient #2874

7.2.3 Shaped Transient Calculations Summary

Table 7.6 summarizes the shaped transient TREKIN calculations, and the results are compared with the measurements.

Table [7.6] Summary of TREKIN Calculations and Measurements for the Shaped Transient #2874

Parameter	Measured (M)	TREKIN Historic	Calc/M	TREKIN MCNP	Calc/M
Max Power (MW)*	262	237	0.90	249	0.95
Core Energy at 20s (MJ)*	1737	1650	0.95	1883	1.08
Max Rod Withdrawal**	94%	97%	1.04	86%	0.92
Negative Reactivity Feedback	n/a	5.63%	n/a	5.28%	n/a

*Rod driven TREKIN calculation

**Power-driven TREKIN calculation

TREKIN underestimates the rod withdrawal using the historic and the MCNP-produced tables by 3% and 8%, respectively due to the lower temperature reactivity calculated with MCNP. Specifically TREKIN calculates 1.07 times lower reactivity feedback when using the MCNP-calculated tables compared to the historic data. As discussed above, the temperature reactivity feedback as a function of total core energy was calculated with MCNP using a temperature distribution based on a constant relative power distribution with control/shutdown rods at a constant position (the transient and compensation rods were fully withdrawn). During the preparation of this report there was no information on how the “historic” temperature reactivity dataset was created.

8. Summary - Conclusions

The aim of this study was to benchmark the MCNP/TREKIN transient calculations for the TREAT HEU core. The method was based on MCNP calculations of the relative power and resulting temperature distribution, approximating the heating of the fuel during a transient as an adiabatic process (the energy deposition is assumed to happen instantaneously). The measured heat capacity of the fuel graphite was used to calculate the temperature of every fuel assembly for several total core energy values. Three radial and three axial temperature zones were defined in the MCNP model by assigning zone-average temperature-dependent cross section libraries created with the MCNP utility program makxsf. This program extrapolates the cross sections libraries and $S(\alpha, \beta)$ thermal scattering data from the already-available cross section libraries. The data tables required by the point kinetics program TREKIN to simulate temperature-limited transients were populated with MCNP-computed values. These MCNP-calculated parameters were the prompt neutron generation lifetime, the effective delayed neutron fraction, the structure of the six delayed neutron groups, the temperature reactivity feedback as a function of total core energy, and the peak and average core fuel temperature as functions of total core energy.

Three experimentally-performed temperature-limited transients were used as benchmarks for the MCNP/TREKIN calculations. TREKIN using the MCNP-calculated input tables overestimated the peak core temperatures by 5% during temperature limited transients. It should be noted that the measurement uncertainties are not known. The experimentally-performed shaped transient #2874 was analyzed with TREKIN in rod-driven and power-driven modes. When using the MCNP-calculated input tables, artifact power peaks were observed in the power-time history calculated in rod-driven TREKIN calculations. The MCNP-calculated temperature reactivity feedback vs. energy is lower than the 'historic' one, and it is suspected that the power is more sensitive in the rod withdrawal fluctuations. This assumption will be evaluated in the sensitivity analysis that is underway.

Based on the findings of this report, the same calculation methods will be used to analyze potential low-enriched uranium (LEU) TREAT cores and evaluate their transient capabilities. Since there will be no near-term measurement data available for the LEU core, evaluation of the relative changes from HEU to LEU will be solely based on the MCNP/TREKIN analyses. The MCNP-computed values of peak core temperature as a function of core energy (which are used as a lookup table in TREKIN) should be used with caution, taking into consideration all the possible sources of uncertainty in the calculations.

Using MCNP/TREKIN calculated power-time histories, the multi-physics code COMSOL is also being used to calculate the peak fuel and cladding temperatures for both HEU and LEU TREAT cores and to analyze the cooling times, including the heating of the radial and axial graphite reflectors. Preliminary analysis of the peak core temperature using COMSOL presented an excellent agreement with the adiabatic temperature calculations during transients presented in this report.

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Appendix A: TREKIN Input Tables

Historic Tables as Used by TREKIN_1.1

!-----
!Transient Rod Bank Pos. (cm.) vs. Reactivity For 1/2 Slot
!Historic Data from plot of transient control rod worths from
!R.W. Swanson, unpublished info (1993). Cubic spline
!interpolation to get additional points (05/18/2012)
!-----

PARAMETER (LenTROD=52)

TABLE TROD,1,LenTROD/

0.0,	2.0,	4.0,	6.0,	8.0,
10.0,	12.0,	14.0,	16.0,	18.0,
20.0,	22.0,	24.0,	26.0,	28.0,
30.0,	32.0,	34.0,	36.0,	38.0,
40.0,	42.0,	44.0,	46.0,	48.0,
50.0,	52.0,	54.0,	56.0,	58.0,
60.0,	62.0,	64.0,	66.0,	68.0,
70.0,	72.0,	74.0,	76.0,	78.0,
80.0,	82.0,	84.0,	86.0,	88.0,
90.0,	92.0,	94.0,	96.0,	98.0,
100.0,	101.6,			
-0.08459,	-0.08327,	-0.08187,	-0.08038,	-0.07882,
-0.07718,	-0.07547,	-0.07369,	-0.07185,	-0.06994,
-0.06798,	-0.06596,	-0.06389,	-0.06178,	-0.05962,
-0.05744,	-0.05523,	-0.05299,	-0.05074,	-0.04849,
-0.04622,	-0.04396,	-0.04171,	-0.03947,	-0.03725,
-0.03505,	-0.03289,	-0.03077,	-0.02869,	-0.02665,
-0.02467,	-0.02274,	-0.02088,	-0.01908,	-0.01735,
-0.01569,	-0.01411,	-0.01260,	-0.01117,	-0.00982,
-0.00856,	-0.00737,	-0.00627,	-0.00525,	-0.00432,
-0.00346,	-0.00269,	-0.00199,	-0.00137,	-0.00082,
-0.00034,	0.00000,			

!-----
!Reactivity vs. Transient Rod Bank Pos. (CM) For 1/2 Slot
!Historic Data (inverse of TROD table) (05/18/2012)
!-----

PARAMETER (LenREVSCM=52)

TABLE RE_VS_CM,1,LenREVSCM/

-0.08459,	-0.08327,	-0.08187,	-0.08038,	-0.07882,
-0.07718,	-0.07547,	-0.07369,	-0.07185,	-0.06994,
-0.06798,	-0.06596,	-0.06389,	-0.06178,	-0.05962,
-0.05744,	-0.05523,	-0.05299,	-0.05074,	-0.04849,
-0.04622,	-0.04396,	-0.04171,	-0.03947,	-0.03725,
-0.03505,	-0.03289,	-0.03077,	-0.02869,	-0.02665,
-0.02467,	-0.02274,	-0.02088,	-0.01908,	-0.01735,
-0.01569,	-0.01411,	-0.01260,	-0.01117,	-0.00982,
-0.00856,	-0.00737,	-0.00627,	-0.00525,	-0.00432,
-0.00346,	-0.00269,	-0.00199,	-0.00137,	-0.00082,
-0.00034,	0.00000,			
0.0,	2.0,	4.0,	6.0,	8.0,
10.0,	12.0,	14.0,	16.0,	18.0,
20.0,	22.0,	24.0,	26.0,	28.0,
30.0,	32.0,	34.0,	36.0,	38.0,
40.0,	42.0,	44.0,	46.0,	48.0,
50.0,	52.0,	54.0,	56.0,	58.0,
60.0,	62.0,	64.0,	66.0,	68.0,
70.0,	72.0,	74.0,	76.0,	78.0,
80.0,	82.0,	84.0,	86.0,	88.0,

```

    90.0,  92.0,  94.0,  96.0,  98.0,
    100.0, 101.6/
!-----
! Source document for HSLLOT, TEMPER and AVGTEMP Tables appears in
! unpublished information from N. A. Kramer and R. H. Dodge,
! (ANL-TREAT) dated 1992.
!-----
! HSLLOT is TREAT Calculation for Half-Slotted Core 8/25/92
!      ENERGY(MJ) vs REACTIVITY
!-----
PARAMETER (LenHSLLOT=51)
TABLE HSLLOT,1,LenHSLLOT/
    0.0, 100.0, 200.0, 300.0, 400.0, 500.0,
    600.0, 700.0, 800.0, 900.0, 1000.0, 1100.0,
    1200.0, 1300.0, 1400.0, 1500.0, 1600.0, 1700.0,
    1800.0, 1900.0, 2000.0, 2100.0, 2200.0, 2300.0,
    2400.0, 2500.0, 2600.0, 2700.0, 2800.0, 2900.0,
    3000.0, 3100.0, 3200.0, 3300.0, 3400.0, 3500.0,
    3600.0, 3700.0, 3800.0, 3900.0, 4000.0, 4100.0,
    4200.0, 4300.0, 4400.0, 4500.0, 4600.0, 4700.0,
    4800.0, 4900.0, 5000.0,
    0.000000, 0.005465, 0.010724, 0.015570, 0.019838, 0.023796,
    0.027477, 0.030844, 0.034160, 0.037392, 0.040304, 0.043094,
    0.045726, 0.048234, 0.050592, 0.052924, 0.055230, 0.057447,
    0.059548, 0.061559, 0.063454, 0.065316, 0.067035, 0.068737,
    0.070424, 0.072096, 0.073753, 0.075396, 0.077025, 0.078640,
    0.080231, 0.081707, 0.083144, 0.084571, 0.085987, 0.087392,
    0.088765, 0.090096, 0.091418, 0.092729, 0.094031, 0.095277,
    0.096477, 0.097632, 0.098780, 0.099919, 0.101051, 0.102176,
    0.103292, 0.104411, 0.105522/
!-----
!   2874 Avg temperature calculated 08/25/92
!-----
PARAMETER (LenAVGTEMP=51)
TABLE AVGTEMP,1,LenAVGTEMP/
    0.0, 100.0, 200.0, 300.0, 400.0, 500.0, 600.0, 700.0,
    800.0, 900.0, 1000.0, 1100.0, 1200.0, 1300.0, 1400.0, 1500.0,
    1600.0, 1700.0, 1800.0, 1900.0, 2000.0, 2100.0, 2200.0, 2300.0,
    2400.0, 2500.0, 2600.0, 2700.0, 2800.0, 2900.0, 3000.0, 3100.0,
    3200.0, 3300.0, 3400.0, 3500.0, 3600.0, 3700.0, 3800.0, 3900.0,
    4000.0, 4100.0, 4200.0, 4300.0, 4400.0, 4500.0, 4600.0, 4700.0,
    4800.0, 4900.0, 5000.0,
    26.2, 44.4, 64.4, 83.4, 100.4, 117.5, 133.6, 148.7,
    163.7, 178.5, 192.1, 205.6, 219.2, 232.3, 244.7, 257.1,
    269.6, 281.7, 293.3, 304.8, 316.4, 327.9, 338.6, 349.3,
    360.0, 370.8, 381.5, 392.2, 402.9, 413.7, 424.3, 434.5,
    444.4, 454.3, 464.3, 474.2, 484.2, 494.1, 504.0, 514.0,
    523.9, 533.5, 542.9, 552.3, 561.7, 571.1, 580.5, 590.0,
    599.4, 608.8, 618.2/
!-----
!   2874 HEU Max temperature calculated 08/25/92
!-----
PARAMETER (LenTEMPER=51)
TABLE TEMPER,1,LenTEMPER/
    0.0, 100.0, 200.0, 300.0, 400.0, 500.0, 600.0, 700.0,
    800.0, 900.0, 1000.0, 1100.0, 1200.0, 1300.0, 1400.0, 1500.0,
    1600.0, 1700.0, 1800.0, 1900.0, 2000.0, 2100.0, 2200.0, 2300.0,
    2400.0, 2500.0, 2600.0, 2700.0, 2800.0, 2900.0, 3000.0, 3100.0,
    3200.0, 3300.0, 3400.0, 3500.0, 3600.0, 3700.0, 3800.0, 3900.0,
    4000.0, 4100.0, 4200.0, 4300.0, 4400.0, 4500.0, 4600.0, 4700.0,
    4800.0, 4900.0, 5000.0,

```

```

27.9, 60.8, 94.3, 125.3, 152.9, 179.9, 204.5, 229.0,
251.7, 274.3, 295.6, 316.6, 336.9, 356.4, 375.9, 395.4,
414.9, 433.9, 452.0, 470.0, 488.1, 506.2, 524.3, 541.6,
558.7, 575.8, 593.0, 610.1, 627.2, 643.3, 659.5, 675.6,
691.8, 707.9, 724.1, 740.2, 756.3, 772.5, 788.6, 804.8,
820.9, 836.5, 851.8, 867.0, 882.2, 897.5, 912.7, 928.0,
943.2, 958.5, 973.7/
!-----
! PHYSICS CONSTANTS NORMALLY DO NOT CHANGE
! TREKIN HISTORIC
! D.Okrent, C.E.Dickerman, J. Gasidlo, D.M. O'Shea, and D. F.
! Schoeberle,"The Reactor Kinetics of the Transient Reactor Test"
! Facility (TREAT), ANL-6174, Sept. 1960.
!
! Page 31 of 1 gave ALAMBD=l=9.0E-4 s
! Page 42 of 1 gave Effective Delayed-neutron fractions B(i),i=1,6,
!     BETA and Decay Constants XL(i),i=1,6
!
!-----
BETA=0.007178
ALAMBD=9.0E-04
CONSTANT B=0.000244,0.001567,0.001411,0.002828,0.000826,0.000302
CONSTANT XL=0.0124,0.0305,0.111,0.3010,1.130,3.00

```

MCNP Calculated Tables as Used by TREKIN_1.1

```

!-----
!Transient Rod Bank Pos. (cm.) vs. Reactivity For 1/2 Slot
! ***** 11 March 2014 *****
! MCNP produced data for HEU 59% from 2855 crit with trans OUT
! multiplied the calculated worth with 0.92
!-----
PARAMETER (LenTROD=9)
TABLE TROD,1,LenTROD/
    0.0, 12.7, 25.4, 38.1, 50.8,
    63.5, 76.2, 88.9, 101.6,
    -0.08367, -0.07436, -0.06108, -0.04784, -0.03406,
    -0.02086, -0.01104, -0.00418, 0.00000/
!-----
!Reactivity vs. Transient Rod Bank Pos. (CM) For 1/2 Slot
! ***** 11 March 2014 *****
!MCNP HEU 59% (inverse of TROD table) (03/11/2014)
!-----
PARAMETER (LenREVSCM=9)
TABLE RE_VS_CM,1,LenREVSCM/
    -0.08367, -0.07436, -0.06108, -0.04784, -0.03406,
    -0.02086, -0.01104, -0.00418, 0.00000,
    0.0, 12.7, 25.4, 38.1, 50.8,
    63.5, 76.2, 88.9, 101.6/
!-----
! HSLLOT is for Half-Slotted Core
! ***** 17 April 2014 *****
! Assuming 193.7MeV per fission for tally normalization
! ENERGY(MJ) vs REACTIVITY
!-----
PARAMETER (LenHSLLOT=11)
TABLE HSLLOT,1,LenHSLLOT/
    0.0, 100.0, 200.0, 300.0, 400.0, 500.0,
    1000.0, 2000.0, 3000.0, 4000.0, 5000.0,

```

```

0.000000, 0.005971, 0.009735, 0.013529, 0.017743, 0.021343,
0.036279, 0.058678, 0.077768, 0.092907, 0.106032/
!-----
! MCNP HEU Avg temperature calculated 04/17/14
!-----
PARAMETER (LenAVGTEMP=11)
TABLE AVGTEMP,1,LenAVGTEMP/
0.0, 100.0, 200.0, 300.0, 400.0, 500.0, 1000.0, 2000.0,
3000.0, 4000.0, 5000.0,
26.0, 46.3, 64.9, 82.3, 98.7, 114.2, 183.6, 298.7,
397.9, 488.6, 573.8/
!-----
! MCNP HEU Max temperature calculated 04/17/14
!-----
PARAMETER (LenTEMPER=11)
TABLE TEMPER,1,LenTEMPER/
0.0, 100.0, 200.0, 300.0, 400.0, 500.0, 1000.0, 2000.0,
3000.0, 4000.0, 5000.0,
26.0, 61.1, 91.9, 119.9, 145.9, 170.3, 276.9, 452.2,
604.9, 746.8, 882.4/
!-----
! PHYSICS CONSTANTS
! HEU 59% MCNP Calculated (kopts card)
!-----
BETA=0.00680
ALAMBD=8.6E-04
CONSTANT B=0.00022,0.00114,0.00111,0.00311,0.00091,0.00032
CONSTANT XL=0.01249,0.03182,0.10938,0.31699,1.35398,8.63652

```

Appendix B: Clustering in EXCEL

The calculated fuel-average temperatures (T_i , $i=1$ to 338) were arranged on a single column on an EXCEL spreadsheet. The clustering method used in this report features the following steps:

1. The fuel assemblies are separated in three zones (groups), and the average temperature was calculated for each, i.e., temperatures T_A , T_B , and T_C . The separation was done in such way to produce similar standard deviations for each group.
2. The *distance*, $(T_{FA} - T_i)^2$, $i=1,2,3$ of every fuel assembly temperature (T_{FA}) from each of the T_1 , T_2 , and T_3 temperatures is calculated. The minimum distance is selected for every fuel assembly.
3. The overall metric is calculated as the sum of all the minimum distances for every fuel assembly. The overall metric is minimized by selecting a new set of zone temperatures (T_1 , T_2 , and T_3) using the *solver* function of Excel.

Assignments of fuel assemblies to the zones, and determination of the zone temperatures, are iterated until a solution is reached in which overall, for the *set of* assemblies, the distances of the assemblies' temperatures from their respective zone temperature are minimized. As an example, Table B.1 shows the new set of zone temperatures as calculated with the *solver* function for the minimum *sum of minimum distances* in the case of total core energy of 100MJ. Table B.2 shows how the fuel assemblies were assigned in each zone according to the minimum distance.

Table [B.1]: The New Set of Temperatures as Calculated by the Solver Function of Excel for the Minimum Sum of Minimum Distances.

Zone 3 T_3 (°C)	Zone 2 T_2 (°C)	Zone 1, T_1 (°C)	Sum of Minimum Distances
52.4	45.4	38.9	1275.413

Table [B.2]: Assignment of the Fuel Assemblies in Three Zones Based on the Temperature Set Calculated with Excel's Solver Function for 100MJ Total Core Energy.

T_{FA} (°C)	$(T_{FA} - T_1)^2$	$(T_{FA} - T_2)^2$	$(T_{FA} - T_3)^2$	Zone	Minimum Distance
34.8	307.7	110.6	16.6	3	16.56
34.9	307.6	110.6	16.5	3	16.53
35.2	295.8	103.6	13.9	3	13.90
35.2	294.4	102.7	13.6	3	13.58
36.1	264.2	85.2	7.7	3	7.74
36.2	261.2	83.6	7.2	3	7.24
36.3	259.6	82.6	7.0	3	6.97
36.3	259.4	82.5	6.9	3	6.94
36.4	255.5	80.3	6.3	3	6.31
36.4	255.1	80.1	6.2	3	6.25
36.7	245.0	74.5	4.8	3	4.75
36.9	239.1	71.2	4.0	3	3.96
37.0	237.2	70.2	3.7	3	3.72
37.0	236.5	69.8	3.6	3	3.64

37.0	236.1	69.6	3.6	3	3.58
37.0	235.4	69.2	3.5	3	3.50
37.1	235.0	69.0	3.4	3	3.45
37.2	232.1	67.5	3.1	3	3.11
37.3	226.9	64.7	2.5	3	2.53
37.5	223.1	62.6	2.1	3	2.14
37.5	221.8	62.0	2.0	3	2.02
37.7	217.2	59.5	1.6	3	1.60
37.8	213.2	57.4	1.3	3	1.27
37.8	212.4	57.1	1.2	3	1.22
38.0	207.3	54.4	0.9	3	0.86
38.0	207.2	54.3	0.8	3	0.85
38.2	200.6	51.0	0.5	3	0.48
38.2	200.0	50.7	0.4	3	0.45
38.3	199.8	50.6	0.4	3	0.44
38.3	198.3	49.8	0.4	3	0.37
38.5	194.0	47.7	0.2	3	0.21
38.5	193.9	47.7	0.2	3	0.21
38.5	192.7	47.1	0.2	3	0.17
38.6	191.4	46.4	0.1	3	0.13
38.6	191.2	46.3	0.1	3	0.13
38.8	184.5	43.0	0.0	3	0.01
38.8	184.4	43.0	0.0	3	0.01
38.8	183.5	42.5	0.0	3	0.01
39.2	173.6	37.9	0.1	3	0.09
39.4	168.7	35.6	0.2	3	0.23
39.6	163.4	33.2	0.5	3	0.47
39.7	161.5	32.3	0.6	3	0.58
39.7	160.7	32.0	0.6	3	0.63
39.8	159.5	31.4	0.7	3	0.71
39.9	157.2	30.4	0.9	3	0.88
39.9	157.0	30.3	0.9	3	0.89
39.9	155.9	29.8	1.0	3	0.97
40.1	150.7	27.6	1.4	3	1.43
40.1	150.5	27.5	1.4	3	1.45
40.1	149.9	27.2	1.5	3	1.51
40.3	146.7	25.9	1.9	3	1.85
40.5	141.8	23.9	2.4	3	2.45
40.5	141.5	23.7	2.5	3	2.49
40.6	138.7	22.6	2.9	3	2.87
40.6	138.1	22.4	3.0	3	2.95
40.7	137.2	22.0	3.1	3	3.10
40.7	136.9	21.9	3.1	3	3.13
40.8	135.3	21.3	3.4	3	3.38
41.1	128.4	18.6	4.6	3	4.59
41.1	128.2	18.5	4.6	3	4.63
41.2	124.8	17.2	5.3	3	5.30
41.4	121.6	16.0	6.0	3	5.97
41.5	119.6	15.3	6.4	3	6.44
41.5	117.9	14.7	6.8	3	6.84
41.6	115.7	13.9	7.4	3	7.38

41.6	115.6	13.9	7.4	3	7.40
41.8	112.2	12.7	8.3	3	8.30
41.8	111.4	12.5	8.5	3	8.50
41.9	109.5	11.9	9.0	3	9.04
42.0	107.5	11.2	9.6	3	9.63
42.0	107.3	11.1	9.7	3	9.71
42.1	105.6	10.6	10.2	3	10.21
42.3	101.4	9.3	11.6	2	9.29
42.4	100.8	9.1	11.8	2	9.10
42.4	100.4	9.0	11.9	2	8.98
42.4	100.4	9.0	11.9	2	8.97
42.4	100.0	8.9	12.0	2	8.88
42.4	99.5	8.7	12.2	2	8.71
42.5	98.3	8.4	12.7	2	8.36
42.5	97.9	8.3	12.8	2	8.25
42.6	95.1	7.5	13.8	2	7.46
42.7	94.5	7.3	14.1	2	7.29
42.8	92.9	6.8	14.7	2	6.84
42.8	92.5	6.7	14.9	2	6.73
42.9	90.1	6.1	15.8	2	6.10
42.9	89.8	6.0	16.0	2	6.03
43.0	89.0	5.8	16.3	2	5.83
43.0	88.6	5.7	16.5	2	5.72
43.0	87.9	5.5	16.8	2	5.53
43.0	87.6	5.5	16.9	2	5.47
43.1	87.2	5.4	17.1	2	5.36
43.1	86.6	5.2	17.4	2	5.22
43.1	86.4	5.2	17.5	2	5.15
43.1	85.7	5.0	17.8	2	4.99
43.2	84.5	4.7	18.3	2	4.70
43.2	84.2	4.6	18.4	2	4.64
43.2	83.9	4.6	18.6	2	4.57
43.3	82.4	4.2	19.3	2	4.23
43.4	81.2	4.0	19.9	2	3.95
43.4	81.2	4.0	19.9	2	3.95
43.4	80.8	3.9	20.1	2	3.87
43.4	80.3	3.8	20.3	2	3.77
43.4	80.3	3.8	20.3	2	3.76
43.4	80.0	3.7	20.5	2	3.69
43.5	79.5	3.6	20.7	2	3.59
43.6	78.0	3.3	21.5	2	3.27
43.6	77.6	3.2	21.8	2	3.19
43.7	75.9	2.8	22.7	2	2.85
43.7	74.9	2.7	23.2	2	2.66
43.8	74.1	2.5	23.7	2	2.52
43.8	73.0	2.3	24.3	2	2.31
43.9	72.0	2.1	24.9	2	2.14
43.9	71.8	2.1	25.0	2	2.11
44.0	71.1	2.0	25.4	2	1.99
44.0	71.0	2.0	25.4	2	1.98
44.0	70.7	1.9	25.7	2	1.91

44.0	70.3	1.9	25.9	2	1.86
44.0	70.3	1.8	25.9	2	1.85
44.0	69.9	1.8	26.1	2	1.80
44.0	69.7	1.8	26.3	2	1.75
44.1	69.5	1.7	26.3	2	1.73
44.1	69.2	1.7	26.6	2	1.67
44.1	69.0	1.6	26.7	2	1.65
44.1	68.0	1.5	27.3	2	1.49
44.2	67.8	1.5	27.4	2	1.47
44.2	67.3	1.4	27.7	2	1.40
44.2	67.1	1.4	27.9	2	1.36
44.2	67.0	1.4	27.9	2	1.36
44.2	66.7	1.3	28.2	2	1.31
44.3	65.0	1.1	29.3	2	1.08
44.4	64.6	1.0	29.6	2	1.03
44.4	64.2	1.0	29.8	2	0.98
44.4	63.5	0.9	30.3	2	0.90
44.4	63.2	0.9	30.5	2	0.87
44.4	63.2	0.9	30.5	2	0.86
44.4	63.1	0.8	30.6	2	0.85
44.5	63.0	0.8	30.6	2	0.84
44.5	62.6	0.8	30.9	2	0.79
44.6	61.2	0.6	31.9	2	0.64
44.6	60.5	0.6	32.4	2	0.57
44.6	60.0	0.5	32.8	2	0.53
44.7	59.2	0.4	33.4	2	0.45
44.7	59.1	0.4	33.5	2	0.44
44.9	56.7	0.3	35.3	2	0.26
44.9	56.3	0.2	35.6	2	0.23
44.9	55.7	0.2	36.1	2	0.20
45.0	54.5	0.1	37.1	2	0.13
45.1	53.5	0.1	37.9	2	0.08
45.2	52.0	0.0	39.2	2	0.03
45.4	48.8	0.0	42.0	2	0.00
45.4	48.7	0.0	42.2	2	0.00
45.5	47.2	0.0	43.6	2	0.02
45.5	47.1	0.0	43.7	2	0.03
45.6	46.7	0.0	44.0	2	0.03
45.6	46.6	0.0	44.2	2	0.04
45.6	45.9	0.1	44.8	2	0.06
45.7	45.4	0.1	45.3	2	0.08
45.8	44.0	0.1	46.7	2	0.15
45.8	42.9	0.2	47.9	2	0.22
45.9	42.3	0.3	48.6	2	0.27
46.0	41.4	0.3	49.6	2	0.35
46.0	41.3	0.4	49.6	2	0.35
46.0	41.0	0.4	50.0	2	0.39
46.0	40.7	0.4	50.3	2	0.41
46.0	40.3	0.4	50.7	2	0.45
46.1	40.2	0.5	50.9	2	0.47
46.1	40.0	0.5	51.0	2	0.48

46.1	39.7	0.5	51.4	2	0.52
46.1	39.6	0.5	51.6	2	0.54
46.1	39.2	0.6	52.0	2	0.58
46.2	38.7	0.6	52.6	2	0.64
46.3	36.9	0.9	54.8	2	0.90
46.3	36.8	0.9	54.9	2	0.92
46.4	36.4	1.0	55.3	2	0.97
46.4	36.0	1.0	55.8	2	1.05
46.4	35.8	1.1	56.0	2	1.07
46.5	35.0	1.2	57.1	2	1.23
46.5	34.4	1.3	57.9	2	1.34
46.6	33.6	1.5	58.9	2	1.50
46.7	32.5	1.7	60.4	2	1.74
46.7	32.4	1.8	60.5	2	1.76
46.7	32.4	1.8	60.6	2	1.78
46.7	31.9	1.9	61.3	2	1.90
46.7	31.8	1.9	61.3	2	1.90
46.8	31.7	1.9	61.5	2	1.95
46.9	30.5	2.2	63.1	2	2.24
46.9	30.0	2.4	63.9	2	2.39
46.9	29.8	2.4	64.2	2	2.44
46.9	29.8	2.4	64.2	2	2.45
47.1	28.1	2.9	66.7	2	2.95
47.1	28.1	2.9	66.7	2	2.95
47.1	28.0	3.0	66.9	2	2.99
47.1	27.7	3.1	67.4	2	3.10
47.2	26.7	3.4	68.9	2	3.43
47.3	25.9	3.7	70.3	2	3.74
47.3	25.6	3.9	70.8	2	3.86
47.3	25.5	3.9	70.9	2	3.89
47.4	24.8	4.2	72.0	2	4.15
47.4	24.6	4.2	72.4	2	4.25
47.5	23.8	4.6	73.8	2	4.59
47.6	23.3	4.8	74.7	2	4.82
47.6	23.3	4.8	74.8	2	4.84
47.6	22.8	5.1	75.7	2	5.06
47.6	22.8	5.1	75.7	2	5.06
47.6	22.8	5.1	75.7	2	5.07
47.7	21.6	5.6	77.9	2	5.64
47.8	20.8	6.1	79.4	2	6.05
47.8	20.6	6.1	79.7	2	6.14
48.0	19.3	6.9	82.3	2	6.88
48.0	19.0	7.1	83.1	2	7.10
48.2	17.7	7.9	85.9	2	7.93
48.2	17.5	8.1	86.3	2	8.08
48.2	17.4	8.1	86.4	2	8.11
48.3	16.3	8.9	88.9	2	8.88
48.4	16.0	9.1	89.7	2	9.12
48.4	15.6	9.5	90.7	2	9.45
48.5	15.2	9.7	91.5	2	9.72
48.5	15.0	9.9	92.2	2	9.93

48.6	14.7	10.2	92.9	2	10.18
48.6	14.5	10.3	93.4	2	10.33
48.6	14.2	10.6	94.2	2	10.60
48.6	14.1	10.7	94.4	2	10.68
48.8	13.1	11.6	97.0	2	11.56
48.8	13.0	11.6	97.3	2	11.64
48.8	12.9	11.7	97.5	2	11.73
48.8	12.7	12.0	98.3	2	12.00
48.9	11.9	12.8	100.4	1	11.90
49.0	11.6	13.1	101.5	1	11.55
49.1	10.9	13.8	103.3	1	10.93
49.1	10.6	14.1	104.2	1	10.65
49.2	10.2	14.7	105.7	1	10.18
49.2	9.9	15.0	106.5	1	9.94
49.3	9.4	15.7	108.3	1	9.40
49.3	9.4	15.7	108.4	1	9.37
49.4	8.8	16.4	110.2	1	8.83
49.6	7.9	17.8	113.8	1	7.85
49.7	7.1	19.0	116.7	1	7.11
49.7	7.1	19.0	116.8	1	7.09
49.8	7.0	19.2	117.4	1	6.95
49.8	7.0	19.2	117.4	1	6.95
49.8	6.6	19.9	119.0	1	6.57
49.9	6.4	20.1	119.5	1	6.45
49.9	6.0	20.8	121.3	1	6.04
50.0	5.9	21.0	121.7	1	5.95
50.0	5.6	21.7	123.3	1	5.61
50.1	5.4	22.1	124.5	1	5.36
50.1	5.4	22.2	124.5	1	5.36
50.1	5.2	22.5	125.3	1	5.19
50.2	4.9	23.0	126.5	1	4.94
50.2	4.8	23.3	127.3	1	4.80
50.3	4.2	24.8	130.5	1	4.19
50.3	4.2	24.8	130.6	1	4.17
50.4	3.8	25.7	132.7	1	3.80
50.5	3.7	25.9	133.1	1	3.74
50.5	3.4	26.8	135.1	1	3.42
50.6	3.3	27.0	135.7	1	3.32
50.7	3.0	27.9	137.7	1	3.02
50.7	3.0	28.1	138.0	1	2.97
50.7	3.0	28.1	138.1	1	2.95
50.7	2.8	28.6	139.1	1	2.81
51.0	2.0	31.5	145.5	1	1.99
51.1	1.8	32.5	147.5	1	1.76
51.4	0.9	36.9	156.9	1	0.90
51.4	0.9	37.0	157.0	1	0.89
51.5	0.7	38.0	159.1	1	0.74
51.6	0.6	38.7	160.5	1	0.64
51.6	0.6	39.2	161.6	1	0.58
51.6	0.6	39.4	161.9	1	0.56
51.7	0.4	40.5	164.2	1	0.43

51.7	0.4	40.7	164.5	1	0.42
51.7	0.4	40.7	164.5	1	0.42
51.8	0.4	40.8	164.8	1	0.40
51.8	0.4	40.9	165.0	1	0.39
51.9	0.3	42.2	167.6	1	0.28
51.9	0.3	42.3	167.8	1	0.27
51.9	0.2	42.8	168.8	1	0.23
52.0	0.2	43.9	171.0	1	0.16
52.1	0.1	45.3	173.8	1	0.08
52.2	0.0	46.9	176.9	1	0.03
52.4	0.0	49.8	182.4	1	0.00
52.6	0.0	51.8	186.3	1	0.03
52.6	0.0	51.9	186.5	1	0.03
52.6	0.1	52.6	187.8	1	0.05
52.7	0.1	53.1	188.6	1	0.07
52.8	0.1	54.8	192.0	1	0.15
52.8	0.2	55.5	193.2	1	0.18
52.8	0.2	55.8	193.8	1	0.20
52.9	0.2	56.2	194.5	1	0.22
52.9	0.3	56.9	195.9	1	0.27
53.0	0.4	58.1	198.0	1	0.36
53.0	0.4	58.5	198.8	1	0.39
53.1	0.5	59.4	200.5	1	0.47
53.2	0.6	61.1	203.4	1	0.63
53.2	0.7	61.7	204.6	1	0.69
53.3	0.8	62.3	205.7	1	0.76
53.3	0.8	62.6	206.3	1	0.79
53.3	0.8	62.7	206.4	1	0.80
53.4	1.0	64.1	209.1	1	0.97
53.4	1.0	64.7	210.0	1	1.04
53.5	1.2	65.9	212.2	1	1.20
53.5	1.2	66.2	212.7	1	1.24
53.5	1.3	66.4	213.1	1	1.27
53.7	1.7	69.4	218.4	1	1.71
53.7	1.8	69.7	219.0	1	1.76
54.0	2.6	74.7	227.8	1	2.62
54.1	2.9	75.9	229.9	1	2.85
54.1	3.0	76.8	231.5	1	3.04
54.2	3.2	77.8	233.1	1	3.23
54.2	3.3	78.2	233.8	1	3.31
54.3	3.7	80.1	237.2	1	3.72
54.3	3.8	80.3	237.6	1	3.77
54.4	4.0	81.2	239.1	1	3.96
54.4	4.1	81.8	240.1	1	4.10
54.4	4.1	82.0	240.3	1	4.13
54.6	4.8	84.9	245.4	1	4.81
54.6	4.9	85.2	245.8	1	4.87
54.6	5.1	86.0	247.2	1	5.07
54.6	5.1	86.1	247.4	1	5.09
54.7	5.4	87.2	249.3	1	5.37
54.7	5.5	87.6	249.9	1	5.46

54.8	5.7	88.5	251.5	1	5.70
54.8	5.8	89.1	252.4	1	5.83
54.8	5.9	89.4	252.9	1	5.91
54.9	6.1	90.1	254.1	1	6.09
54.9	6.1	90.1	254.2	1	6.10
54.9	6.5	91.5	256.6	1	6.48
55.0	6.8	92.7	258.4	1	6.78
55.0	7.0	93.4	259.6	1	6.97
55.0	7.0	93.5	259.9	1	7.02
55.0	7.0	93.6	260.0	1	7.04
55.1	7.4	94.8	262.0	1	7.36
55.2	7.6	95.7	263.5	1	7.63
55.2	7.9	96.7	265.1	1	7.89
55.2	8.1	97.3	266.1	1	8.07
55.3	8.6	99.1	269.2	1	8.61
55.3	8.7	99.3	269.5	1	8.66
55.4	8.8	99.9	270.4	1	8.84
55.5	9.8	103.2	275.8	1	9.84



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